April 14, 1992

Docket Nos. 50-413
and 50-414

Mr. M. S. Tuckman
Vice President, Catawba Site
Duke Power Company
4800 Concord Road
York, South Carolina 29745

Dear Mr. Tuckman:

SUBJECT: ISSUANCE OF AMENDMENT NO. 95 TO FACILITY OPERATING LICENSE NPF-35 AND AMENDMENT NO. 89 TO FACILITY OPERATING LICENSE NPF-52 - CATAWBA NUCLEAR STATION, UNITS 1 AND 2 (TACS M77339, M77340, M77409, M77410)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 95 to Facility Operating License NPF-35 and Amendment No. 89 to Facility Operating License NPF-52 for the Catawba Nuclear Station, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated May 9, 1991, as supplemented on February 6, 1992.

The amendments revise the the TSs in response to the guidance of Generic Letter 90-06 to enhance the reliability of power operated relief valves (PORVs) and block valves, and to provide additional low-temperature overpressure protection.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/s/

Robert E. Martin, Senior Project Manager
Project Directorate II-3
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 95 to NPF-35
2. Amendment No. 89 to NPF-52
3. Safety Evaluation

cc w/enclosures:
See next page

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3/18/91	3/26/91	07/24/91	03/25/91	3/13/91	4/14/91

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

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cc w/enclosures:
See next page

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Duke Power Company

Catawba Nuclear Station

cc:

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DATED: April 14, 1992

AMENDMENT NO. 95 TO FACILITY OPERATING LICENSE NPF-35 - Catawba Nuclear Station, Unit 1
AMENDMENT NO. 89 TO FACILITY OPERATING LICENSE NPF-52 - Catawba Nuclear Station, Unit 2

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

DUKE POWER COMPANY

NORTH CAROLINA ELECTRIC MEMBERSHIP CORPORATION

SALUDA RIVER ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-413

CATAWBA NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 95
License No. NPF-35

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Catawba Nuclear Station, Unit 1 (the facility) Facility Operating License No. NPF-35 filed by the Duke Power Company, acting for itself, North Carolina Electric Membership Corporation and Saluda River Electric Cooperative, Inc. (licensees) dated May 9, 1991, as supplemented on February 6, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-35 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 95 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Duke Power Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

D B Matthews

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
  Changes

Date of Issuance: April 14, 1992

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

DUKE POWER COMPANY

NORTH CAROLINA MUNICIPAL POWER AGENCY NO. 1

PIEDMONT MUNICIPAL POWER AGENCY

DOCKET NO. 50-414

CATAWBA NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 89
License No. NPF-52

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Catawba Nuclear Station, Unit 2 (the facility) Facility Operating License No. NPF-52 filed by the Duke Power Company, acting for itself, North Carolina Municipal Power Agency No. 1 and Piedmont Municipal Power Agency (licensees) dated May 9, 1991 as supplemented on February 6, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-52 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 89 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Duke Power Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

DBMatthews

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
  Changes

Date of Issuance: April 14, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 95

FACILITY OPERATING LICENSE NO. NPF-35

DOCKET NO. 50-413

AND

TO LICENSE AMENDMENT NO. 89

FACILITY OPERATING LICENSE NO. NPF-52

DOCKET NO. 50-414

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change.

Remove Pages	Insert Pages
3/4 4-10	3/4 4-10
3/4 4-11	3/4 4-11
3/4 4-37	3/4 4-37
3/4 4-38	3/4 4-38
B 3/4 4-2	B 3/4 4-2
B 3/4 4-3	B 3/4 4-3
B 3/4 4-3a	B 3/4 4-3a

REACTOR COOLANT SYSTEM

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 All power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or more PORV(s) inoperable, because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one or two PORVs inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve(s); restore the PORV(s) to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three PORVs inoperable due to causes other than excessive seat leakage, within 1 hour either restore at least one PORV to OPERABLE status or close their associated block valves and remove power from the block valves and be in HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

d. With one or more block valve(s) inoperable and not closed, within 1 hour restore the block valve(s) to OPERABLE status, or place its associated PORV switch(es) in the 'close' position. Restore at least one block valve to OPERABLE status within the next hour if three block valves are inoperable; restore any remaining inoperable block valve(s) to OPERABLE status within 72 hours; otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

e. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by:

a. Performance of a CHANNEL CALIBRATION, and

b. Operating the valve through one complete cycle of full travel*.

4.4.4.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed with power removed in order to meet the requirements of ACTION b. or c. in Specification 3.4.4.

4.4.4.3 The safety related nitrogen supply for the PORVs shall be demonstrated OPERABLE at least once per 18 months by:

a. Manually transferring motive power from the normal (air) supply to the emergency (nitrogen) supply,

b. Isolating and venting the normal (air) supply, and

c. Operating the valves through a complete cycle of full travel.

*In order to simulate environmental effects representative of operating conditions SR 4.4.4.1b should be conducted when the reactor coolant system temperature is greater than 200°F; however this SR shall not be performed in MODES 1 or 2.

LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following Overpressure Protection Systems shall be OPERABLE:

a. Two power operated relief valves (PORVs) with a lift setting of less than or equal to 450 psig, or

b. The Reactor Coolant System depressurized with a Reactor Coolant System vent of greater than or equal to 4.5 square inches.

APPLICABILITY: MODE 4 when the temperature of any Reactor Coolant System cold leg is less than or equal to 285°F, MODE 5 and MODE 6 when the head is on the reactor vessel.

ACTION:

a. With one PORV inoperable in MODE 4, restore the inoperable PORV to OPERABLE status within 7 days or complete depressurization and venting of the Reactor Coolant System through at least 4.5 square inch vent within the next 8 hours.

b. With one PORV inoperable in MODES 5 or 6, restore the inoperable PORV to OPERABLE status within 24 hours or complete depressurization and venting of the Reactor Coolant System through at least 4.5 square inch vent within the next 8 hours.

c. With both PORVs inoperable, complete depressurization and venting of the Reactor Coolant System through at least a 4.5 square inch vent within 8 hours.

d. In the event either the PORVs or the Reactor Coolant System vent(s) are used to mitigate a Reactor Coolant System pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or Reactor Coolant System vent(s) on the transient, and any corrective action necessary to prevent recurrence.

e. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, at least once per 31 days;

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and

c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

4.4.9.3.2 The Reactor Coolant System vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

SAFETY VALVES (Continued)

relief capability and will prevent overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against overpressurization at low temperatures.

During operation, all pressurizer Code safety valves must be OPERABLE to prevent the Reactor Coolant System from being pressurized above its Safety Limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss-of-load assuming no Reactor trip until the first Reactor Trip System Trip Setpoint is reached (i.e., no credit is taken for a direct Reactor trip on the loss-of-load) and also assuming no operation of the power-operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.3 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady-state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12-hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the Reactor Coolant System is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

3/4.4.4 RELIEF VALVES

The power-operated relief valves (PORVs) and steam bubble function to relieve Reactor Coolant System pressure during all design transients up to and including the design step load decrease with steam dump. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable. The OPERABILITY of the PORVs and block valves is determined on the basis of their being capable of performing the following functions: 1) Manual control of PORVs to control Reactor Coolant System pressure. This is a function that is used for the steam generator tube rupture accident coincident with a loss of all offsite power and for plant shutdown. 2) Maintaining the integrity of the reactor coolant pressure boundary. This is a function that is related to controlling identified leakage and ensuring the ability to detect unidentified reactor coolant pressure boundary leakage. 3) Manual control of the block valve to unblock an isolated PORV to allow it to be used for manual control of Reactor Coolant System pressure and isolate a PORV with excessive seat leakage. 4) Automatic control of PORVs to control

STEAM GENERATORS (Continued)

reactor coolant system pressure except for limited periods where the PORV has been isolated due to excessive seat leakage and except for limited periods where the PORV and/or block valve is closed because of testing and is fully capable of being returned to its normal alignment at any time, provided that this evolution is covered by an approved procedure. This is a function that reduces challenges to the code safety valves for overpressurization events. 5) Manual control of a block valve to isolate a stuck-open PORV. Testing of the PORVs includes the emergency N_2 supply from the Cold Leg Accumulators. This test demonstrates that the valves in the supply line operate satisfactorily and that the nonsafety portion of the instrument air system is not necessary for proper PORV operation.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the Reactor Coolant System will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The B&W process (or method equivalent) to the inspection method described in Topical Report BAW-2045(P)-A will be used. Inservice inspection of steam generator sleeves is also required to ensure RCS integrity. Because the sleeves introduce changes in the wall thickness and diameter, they reduce the sensitivity of eddy current testing, therefore, special inspection methods must be used. A method is described in Topical Report BAW-2045(P)-A with supporting validation data that demonstrates the inspectability of the sleeve and underlying tube. As required by NRC for licensees authorized to use this repair process, Catawba commits to validate the adequacy of any system that is used for periodic inservice inspections of the sleeves, and will evaluate and, as deemed appropriate by Duke Power Company, implement testing methods as better methods are developed and validated for commercial use.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Reactor Coolant System and the Secondary Coolant System (reactor-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a reactor-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that reactor-to-secondary leakage of 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and repaired.

STEAM GENERATORS (Continued)

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Repair will be required for all tubes with imperfections exceeding the repair limit of 40% of the tube nominal wall thickness. For Unit 1, defective tubes which fall under the alternate tube plugging criteria do not have to be repaired. Defective steam generator tubes can be repaired by the installation of sleeves which span the area of degradation, and serve as a replacement pressure boundary for the degraded portion of the tube, allowing the tube to remain in service. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect wastage type degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission pursuant to Specification 6.9.2 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary. If a tube is sleeved due to degradation in the F* distance, then any defects in the tube below the sleeve will remain in service without repair.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These Detection Systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 95 TO FACILITY OPERATING LICENSE NPF-35

AND AMENDMENT NO. 89 TO FACILITY OPERATING LICENSE NPF-52

DUKE POWER COMPANY, ET AL.

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

1.0 INTRODUCTION

On June 25, 1990, the Nuclear Regulatory Commission (NRC) issued Generic Letter 90-06, "Resolution Of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,' Pursuant to 10 CFR 50.54(f)." The generic letter represented the technical resolution of the above mentioned generic issues.

Generic Issue 70, "Power-Operated Relief Valve and Block Valve Reliability," involves the evaluation of the reliability of power-operated relief valves (PORVs) and block valves, and their safety significance in PWR plants. The generic letter discussed how PORVs are increasingly being relied on to perform safety-related functions and the corresponding need to improve the reliability of both PORVs and their associated block valves. Proposed staff positions and improvements to the plant's technical specifications were recommended to be implemented at all affected facilities. This issue is applicable to all Westinghouse, Babcock & Wilcox, and Combustion Engineering designed facilities with PORVs.

Generic Issue 94, "Additional Low-Temperature Overpressure Protection for Light-Water Reactors," addresses concerns with the implementation of the requirements set forth in the resolution of Unresolved Safety Issue (USI) A-26, "Reactor Vessel Pressure Transient Protection (Overpressure Protection)." The generic letter discussed the continuing occurrence of overpressure events and the need to further restrict the allowed outage time for a low-temperature overpressure protection channel in operating modes 4, 5, and 6. This issue is only applicable to Westinghouse and Combustion Engineering facilities.

By letter dated May 9, 1991, Duke Power Company, et al. (the licensee), proposed changes to the Catawba Nuclear Station, Units 1 and 2, Technical Specifications in response to Generic Letter 90-06. By letter dated December 18, 1991, the NRC provided clarifications and requested revisions to the licensee's submittal. By letter dated February 6, 1992, the licensee responded with revisions that are consistent with the guidance of GL 90-06. The February 6, 1992, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

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2.1 Evaluation For Generic Issue 70

The actions proposed by the NRC staff to improve the reliability of PORVs and block valves represent a substantial increase in overall protection of the public health and safety and a determination has been made that the attendant costs are justified in view of this increased protection. The technical findings and the regulatory analysis related to Generic Issue 70 are discussed in NUREG-1316, "Technical Findings and Regulatory Analysis Related to Generic Issue 70, 'Evaluation of Power-Operated Relief Valve Reliability in PWR Nuclear Power Plants.'"

The Technical Specification (TS) changes in response to Generic Issue 70, "Power-Operated Relief Valve and Block Valve Reliability," consist of changes to TS 3/4.4.4, Relief Valves. An assessment of the proposed TS against the model TS of Generic Letter (GL) 90-06 for a Westinghouse plant with three PORV's follows.

Action statement a. is changed to require that power be maintained to the block valves when they are closed due to excessive PORV leakage.

Action statements a., b., c., and d. have been modified such that they terminate in HOT SHUTDOWN within six hours of the preceding action instead of terminating in COLD SHUTDOWN within 30 hours of the preceding action.

Action statement b. is changed to include the case where one or two PORVs (versus one before) are inoperable. Action statement c. is changed to require that at least one PORV must be restored, etc., with three PORVs inoperable instead of requiring each PORV to be restored, etc., when more than one is inoperable.

The licensee states that the change submitted for action statement d. deviates slightly from the guidance in the GL in that the action statement only applies when the block valves are inoperable and not closed (per the addition of the phrase "and not closed"). The licensee considers that if the block valves are inoperable while closed, then the PORV flow path itself would be considered to be inoperable, and accordingly, action statement b. or c. would govern the required action.

Action statement d. also deviates from the GL in the directions for positioning of the PORV switches in the event of inoperable block valves(s). The GL guidance was to place the PORV (with an operable block valve) in manual control to preclude its automatic opening and subsequent potential for a stuck-open PORV. The Catawba PORV switches are labeled "open," "close," and "auto" so the license submits that its proposal to place the PORV switches in the "close" position in such circumstances will likewise preclude automatic PORV opening and the subsequent potential for a stuck-open PORV when the block valve is inoperable and not closed.

The licensee's initially proposed Surveillance Requirement (SR) 4.4.4.1 for operating the PORV through one complete cycle of full travel did not include the stipulation that this be done in MODES 3 or 4. The licensee stated that it does stroke the valves during MODE 4, but concludes that it would not be appropriate to include an SR for a MODE 4 action in this TS since the TS's applicability is only to MODES 1, 2, and 3. This was addressed by a letter from the NRC staff dated December 18, 1991. The licensee's response dated February 6, 1992, indicates that this testing will be done at temperatures greater than 200°F which is consistent with entry into MODE 4 conditions. This is an acceptable response to this concern.

The licensee's proposed SR 4.4.4.1 does not require operating the solenoid air control valves and check valves on accumulators in PORV control systems through a complete cycle of full travel. This is because the action required by SR 4.4.4.3., fully stroking the PORVs while aligned to the emergency nitrogen supply, cycles the necessary valves. Therefore, the licensee did not expand SR 4.4.4.1 to include this requirement.

The guidance contained in the GL for SR 4.4.4.3 indicates that motive and control power for the PORVs and block valves should be manually transferred from the normal to the emergency power bus. This would be directly applicable to a design wherein non-safety related electrical power supplies for both motive and control power are provided for these valves. However, the Catawba PORVs are air operated; the block valves are electrically powered from an essential (emergency or safety related) bus, and control power is from essential sources for the PORV and the block valves. The Catawba SR 4.4.4.3, as currently written, appropriately addresses the PORV motive power transfer from normal (air) to the emergency (nitrogen) supply to demonstrate operability of the emergency nitrogen supply. Since the block valves' motive and control power is normally from essential electrical power, their inclusion in SR 4.4.4.3 is extraneous and the licensee has proposed its removal from SR 4.4.4.3.

The NRC staff has reviewed the licensee's proposed modifications to the Catawba Nuclear Station Technical Specifications. Since the proposed modifications are consistent with the staff's position previously stated in the GL and found to be justified in the above mentioned regulatory analysis, the staff finds the proposed modifications to be acceptable.

The licensee has also expanded the BASES Section 3/4.4.4 to identify the major function of the PORVs and block valves as follows:

1) Manual control of Reactor Coolant System pressure following accidents,

2) Maintaining reactor coolant pressure boundary integrity by controlling leakage,

3) Manual control of block valves to isolate and unblock PORVs (for manual pressure control and for controlling PORV seat leakage),

4) Automatic control of Reactor Coolant System pressure, except for limited periods when the PORV has been isolated due to excessive seat leakage and except for limited periods where the PORV and/or block valve is closed because of testing and is fully capable of being returned to its normal alignment at any time, provided that this evolution is covered by an approved procedure. This is a function that reduces challenges to the code safety valves for overpressurization events.

5) Manual control of block valves to isolate a stuck-open PORV.

These expanded BASES are consistent with the guidance of GL 90-06.

2.2 Evaluation For Generic Issue 94

The actions proposed by the NRC staff improve the availability of the low-temperature overpressure protection (LTOP) system represents a substantial increase in the overall protection of the public health and safety and a determination has been made that the attendant costs are justified in view of this increased protection. The technical findings and the regulatory analysis related to Generic Issue 94 are discussed in NUREG-1326, "Regulatory Analysis for the Resolution of Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors.'"

The TS changes in response to Generic Issue 94, "Additional Low-Temperature Overpressure Protection for Light Water Reactors," include changes to TS 3/4.4.9.3, "Overpressure Protection Systems." An assessment of the proposed TS against the model TS of GL 91-06 for a Westinghouse plant follows.

The licensee notes that the GL TS proposes that the APPLICABILITY of the Limiting Condition for Operating (LCO) for TS 3.4.9.3 be changed to exclude MODE 6 when the Reactor Coolant System (RCS) is adequately vented and that the depressurizing and venting of the RCS not be classified as an overpressure protection system. The GL also proposes that an additional action statement be added to specify verifying the vent pathway when the RCS is depressurized and vented. The licensee concludes that this proposed structure appears inappropriate, because once the RCS is vented, LCO 3.4.9.3. would no longer apply and the action statement requiring verification of the vent pathway would, therefore, not have to be entered. For this reason, the licensee proposed that the present structure of the Catawba TS be maintained in that the depressurizing and venting of the RCS will continue to be classified as an overpressure protection system and the requirement to verify the vent pathway when the system is depressurized and vented will continue to be governed by SR 4.4.9.3.2.

The NRC staff has considered the licensee's proposal and agrees with it since it would not be consistent with the intent of the GL to fail to verify the vent pathway when the vent is being used for overpressure protection. Therefore, the TSs proposed by the licensee in its item 3.4.9.3.b and 4.4.9.3.2 are acceptable.

The licensee proposes to change the language of the APPLICABILITY statement from "...with the reactor vessel head on." to "...when the head is on the reactor vessel.," consistent with the language of the GL. This is acceptable.

Action statement a. is proposed to be modified to clarify that it is only applicable in MODE 4. This is consistent with the guidance in the GL and is acceptable.

Action statement b. is added to reduce the allowable outage time for an inoperable PORV in MODES 5 or 6 from 7 days to 24 hours. This is consistent with a key position of GL 90-06 for the resolution of Generic Issue 94 and is acceptable.

Action statement a., new statement b., and renumbered statement c. are clarified by inclusion of the words "...complete depressurization and venting of..." in lieu of "...depressurize and vent..." This clarifies that these actions must be completed within the specific period. This clarification proposed by the licensee is acceptable.

The licensee proposes to simplify SR 4.4.9.3 by removing requirements that exist because of general requirements applicable to all surveillance requirements as specified in Section 4.0 of the TS. This is consistent with GL 90-06 guidance and is acceptable.

The NRC staff has reviewed the licensee's proposed modifications to the Catawba Nuclear Station Technical Specifications. Since the proposed modifications are consistent with the staff's position previously stated in the generic letter and justified in the above mentioned regulatory analysis, the staff finds the proposed modifications to be acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the South Carolina State official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (56 FR 31433). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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